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ABSTRACT

This seventh in a series of eight modules for a course titled Mechanical Inspection is devoted to the design and fabrication of the reactor pressure vessel. The module follows a typical format that includes the following sections: (1) introduction, (2) module prerequisites, (3) objectives, (4) notes to instructor/student, (5) subject matter, (6) materials needed, (7) procedures, (8) data table, (9) reference materials, (10) test. (The course is one of fifteen technical courses for the second year of a two-year postsecondary curriculum in Nuclear Quality Assurance/Quality Control Technology.) (LRA)

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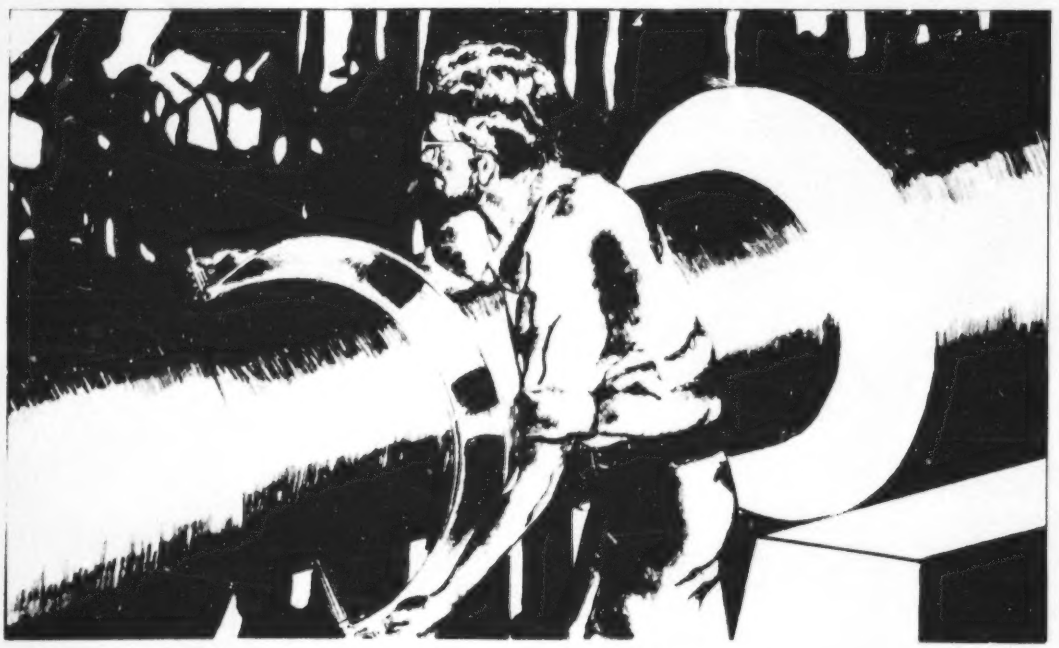
NUCLEAR TECHNOLOGY

COURSE 30 MECHANICAL INSPECTION

MODULE 30-7 Pressure Vessel Inspection



CE 024 283

GENERAL PHYSICS CORPORATION
Columbia , Maryland

COURSE MODULES

COURSE 30 MECHANICAL INSPECTION

Module 30-1	Pump Inspection
Module 30-2	Pump Functional Testing
Module 30-3	Valve Inspection
Module 30-4	Piping Inspection
Module 30-5	Surface Cleaning Inspection
Module 30-6	Protective Coating Inspection
Module 30-7	Pressure Vessel Inspection
Module 30-8	Rotating Equipment Inspection

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This module was prepared for General Physics Corporation by Chet Kupiec and John Espy, General Physics Corporation.

GENERAL PHYSICS CORPORATION

1979

INTRODUCTION

The reactor pressure vessel can be considered the heart of the reactor coolant system for a nuclear power plant.

The heat generated by the fission process in the nuclear fuel is utilized to create steam to drive the turbine-generator. Because of the high pressures and temperatures to which the reactor pressure vessel is subjected, great care must be exercised in its design and fabrication. Heat-up and cooldown effects during plant startup and shutdown cause great stresses on the reactor pressure vessel. The reactor operator must follow operating procedures closely to ensure that the design limitations on this pressure vessel are never exceeded.

This module is devoted to the design and fabrication of the reactor pressure vessel. Although there are other smaller pressure vessels, in various systems in a nuclear generating station, the one vessel of major concern insofar as the resultant consequences of its possible failure is the reactor pressure vessel or simply reactor vessel.

The potential catastrophic failure of the reactor vessel has necessitated rigid standards for its design, fabrication, testing and quality control. Destructive testing is employed to establish the material properties and limits which the fabrication material can withstand. In this manner, it is determined if the material can satisfactorily withstand the stresses and rigors imposed upon an operating reactor vessel. Nondestructive testing is employed to determine the existence and location of flaws in the vessel material and in the welds that join different sections of the vessel together. By thus limiting the number and size of flaws or discontinuities, one can reduce the probability of vessel catastrophic failure.

For purposes of illustration in this module, a pressurized water reactor (PWR) vessel is considered. However, it should be realized that the fabrication and testing requirements for a boiling water reactor (BWR) vessel are somewhat less restrictive than those for a PWR due to its lower operating pressure and temperature. However, the design, fabrication and testing requirements for both PWR and BWR vessels are very similar.

MODULE PREREQUISITES

The prerequisites for this module are:

1. Prior study or experience in basic metallurgy and metals properties (Example: Nuclear Technology Course #15, Metallurgy and Metals Properties).
2. Prior study or experience in nondestructive examinations (Example: Nuclear Technology Course #26, NDE Techniques I).

OBJECTIVES

Upon completion of this module, the student/trainee can:

1. State the primary reason for application of stringent requirements to the design, fabrication, installation, and operation of reactor vessels.
2. Recognize sources of stresses on the reactor vessel in operation and correlate the significance of these sources with ¼ and 3/4 thickness locations.
3. Define elasticity and brittle fracture.

4. Define T_{NDT} and RT_{NDT}.
5. Describe how T_{NDT} and RT_{NDT} are determined.
6. State the purpose of chemistry analysis and tensile tests of materials.
7. Describe the effects of neutron irradiation of the reactor vessel on RT_{NDT}.
8. Describe the basis for the establishment of heatup and cooldown limits for reactor operation.
9. Name the documents used for procurement of reactor vessels.
10. State the purpose of volumetric nondestructive examination of materials used for fabrication of reactor vessels.
11. Briefly describe controls applicable to weld repair of flaws in materials.
12. Describe certification of material by the material manufacturer.
13. Describe marking of materials.
14. State three purposes of fabrication processing.
15. Describe typical fabrication controls by indicating the nature of controls applied to (a) material identity, (b) dimensions, (c) welding, and (d) nondestructive testing.
16. State the purpose of reactor vessel hydrostatic testing and briefly describe conduct of hydrostatic testing.
17. Describe the reactor vessel certification by the fabricator.
18. State rationale for source inspection of reactor vessels by the nuclear utility.

NOTES TO INSTRUCTOR / STUDENT

Due to the very specific nature of this module and because of the detailed code requirements discussed, it would be very useful for the student/trainee to have access to Section III, Subsections NA and NB, of the ASME Boiler and Pressure Vessel Code for additional reference. A brief review of metallurgy and metals properties and non-destructive examination would also be beneficial for a more complete understanding of reactor pressure vessel construction and inspection.

SUBJECT MATTER

FUNDAMENTAL CONCEPTS

Need for Mechanical Inspection of Pressure Vessels

The most serious accident which could occur at a nuclear generating station is the catastrophic failure of the reactor vessel. Such failure makes cooling of the reactor fuel extremely difficult because a level of water probably could not be sustained in the reactor vessel. Therefore, stringent standards of testing and quality control are applied to the design, fabrication, and installation of reactor vessels.

Reactor Vessel Functions

The reactor vessel is a highly complex pressure vessel which has a number of functions in operation of a nuclear power plant. Of the two basic types, pressurized water reactors (PWR) and boiling water reactors (BWR), the reactor vessel serves essentially the same functions. These are:

- serve as pressure retaining vessel
- provide physical support to the reactor core
- receive inlet and outlet cooling water for PWRs and recirculating water for BWRs
- for BWRs, receive feed water and discharge steam
- provide physical support for control rod mechanisms
- provide access to reactor core for refueling
- receive instrumentation sensing lines and detectors.

A typical reactor vessel for a PWR is illustrated in Figure 30-7-1. Examination of Figure 30-7-1 reveals several important concepts. Stresses are not uniformly distributed and material thicknesses vary in order to provide sufficient strength based on stresses seen in operation. Internal surfaces are clad with corrosion resistant material. Location and orientation of nozzles and penetrations

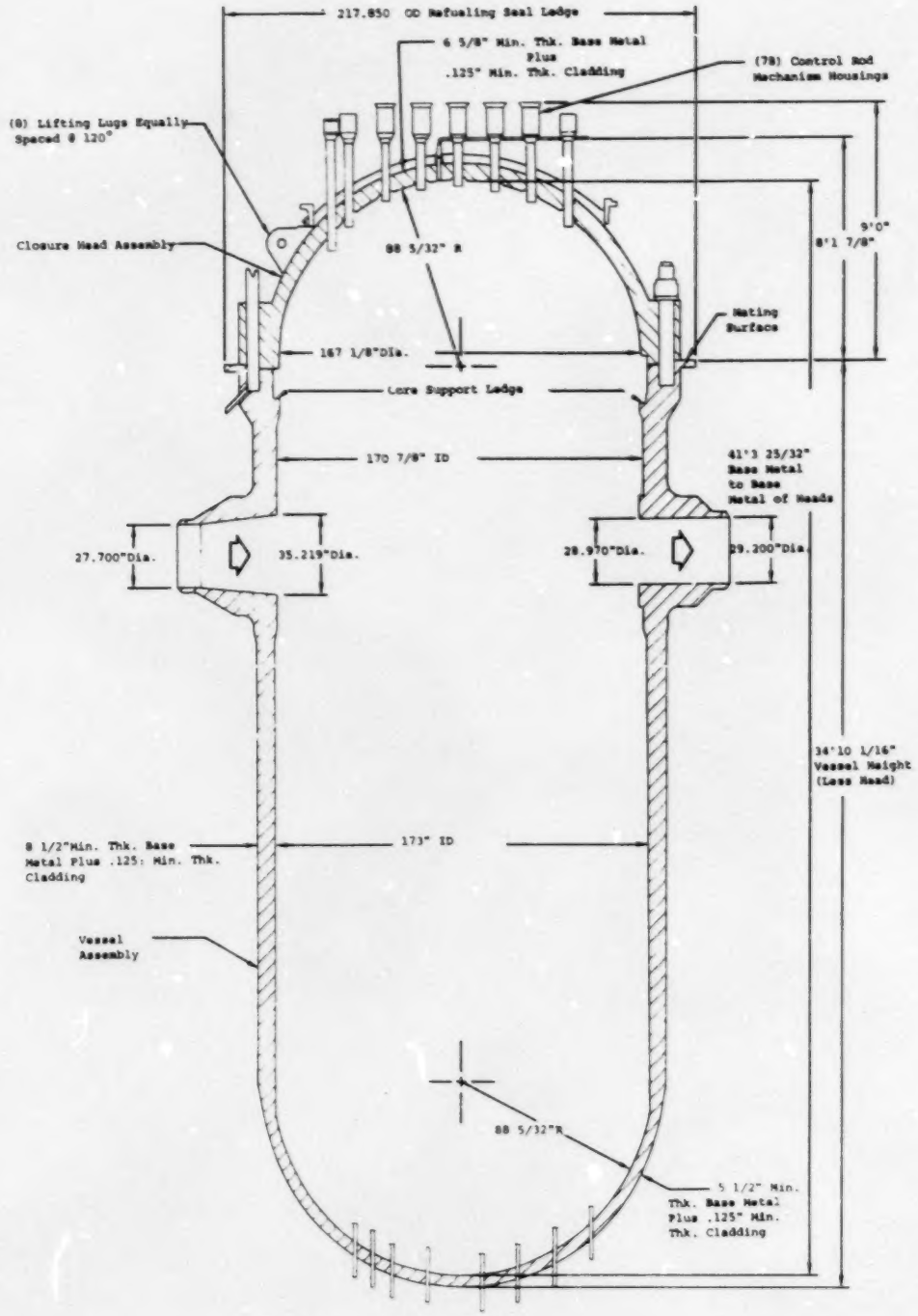

217.650 OD Refueling Seal Ledge
6 5/8" Min. Thk. Base Metal Plus .125" Min. Thk. Cladding
(78) Control Rod Mechanism Housings
(8) Lifting Lugs Equally Spaced @ 120°
Closure Head Assembly
88 5/32" R
9'0"
8'1 7/8"
Mating Surface
167 1/8"Dia.
Core Support Ledge
170 7/8" ID
41'3 25/32" Base Metal to Base Metal of Heads
27.700"Dia.
35.219"Dia.
28.970"Dia.
29.200"Dia.
34'10 1/16" Vessel Height (Less Head)
8 1/2"Min. Thk. Base Metal Plus .125: Min. Thk. Cladding
173" ID
Vessel Assembly
88 5/32"R
5 1/2" Min. Thk. Base Metal Plus .125" Min. Thk. Cladding

Figure 30-7-1. Typical Reactor Vessel for a PWR

must be compatible with connecting piping and tubing. A bolted closure head assembly provides for access for refueling. Large dimensions are involved. The reactor vessel receives a relatively large neutron and gamma radiation flux during power operation. Thermally induced stresses are important due to the relatively large material thicknesses. Except for the closure head bolting, an all welded construction is used. Material and weld joint strength and ductility are important in order to avoid failure. Soundness of materials and welds is important in order to avoid failure.

Stresses Developed in the Reactor Vessel Wall

Stresses arising from coolant system pressure exerted against the vessel wall are always tensile in nature. Stresses arising from temperature gradients across the vessel wall can either be tensile or compressive. The type of stress is a function of the wall thickness and reverses from heatup to cooldown. During system heatup, the vessel outer wall temperature lags the inner wall temperature. The stresses produced by this temperature gradient and by system pressure produce the profile of Figure 30-7-2.

It can be seen that while the pressure stresses are always tensile, at the 1/4 thickness (1/4T), the temperature stresses are compressive. Thus the stresses at the 1/4 T location tend to cancel during system heatup. At the 3/4 T location, however, the stresses from both temperature and pressure are tensile and thus reinforce each other during system heatup. For this reason the 3/4 T location is limiting during system heatup.

During system cooldown, the stress profile of Figure 30-7-3 is obtained. During cooldown, the temperature of the outer wall lags the temperature of the inner wall and is at a higher temperature. It can be seen that during

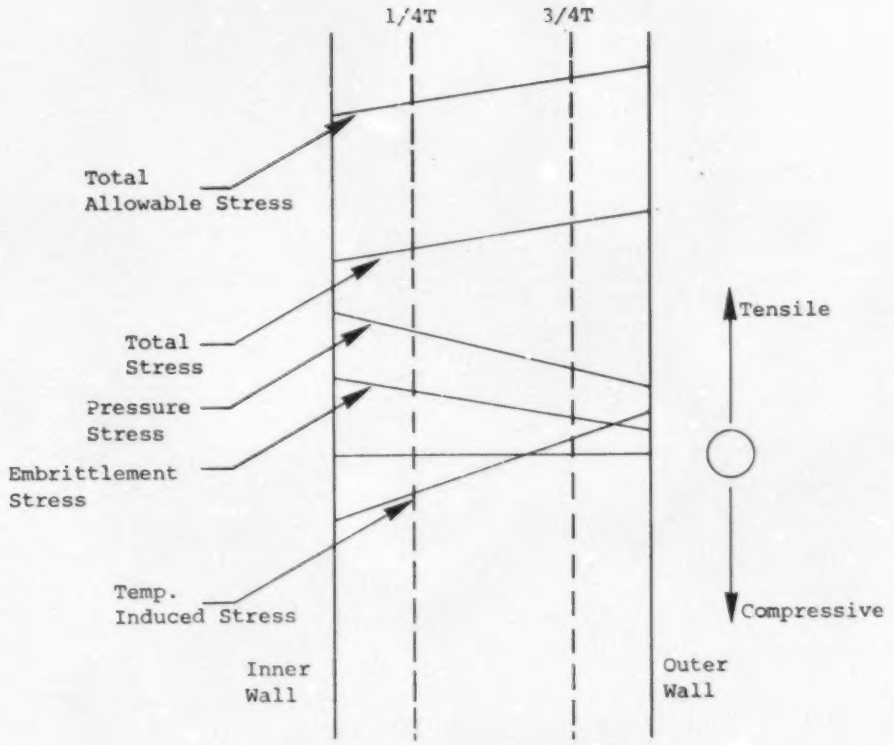


Figure 30-7-2. Heatup Stress Profile

cooldown, the stresses at the 3/4 T location are tensile due to system pressure and compressive due to the temperature gradient. Thus during cooldown, the stresses at the 3/4T location tend to cancel as shown in Figure 30-7-3. At the 1/4 T location, however, the pressure and temperature stresses are both tensile and thereby reinforce each other. Thus the 1/4 T location is limiting during system cooldown.

Neutron irradiation of the vessel wall produces damage that makes it more brittle. For this reason, an embrittle-

ment stress component must be included in the stress profile of the vessel wall. The time integrated neutron dose (fluence) is greater at the 1/4 T location than at the 3/4 T location, as shown in Figure 30-7-4. For this reason, the total stress realized during system cooldown more closely approaches the total allowable stress. Because system cooldown is a more limiting situation, the 1/4 T value is used for heatup and cooldown rate calculations because it is more conservative.

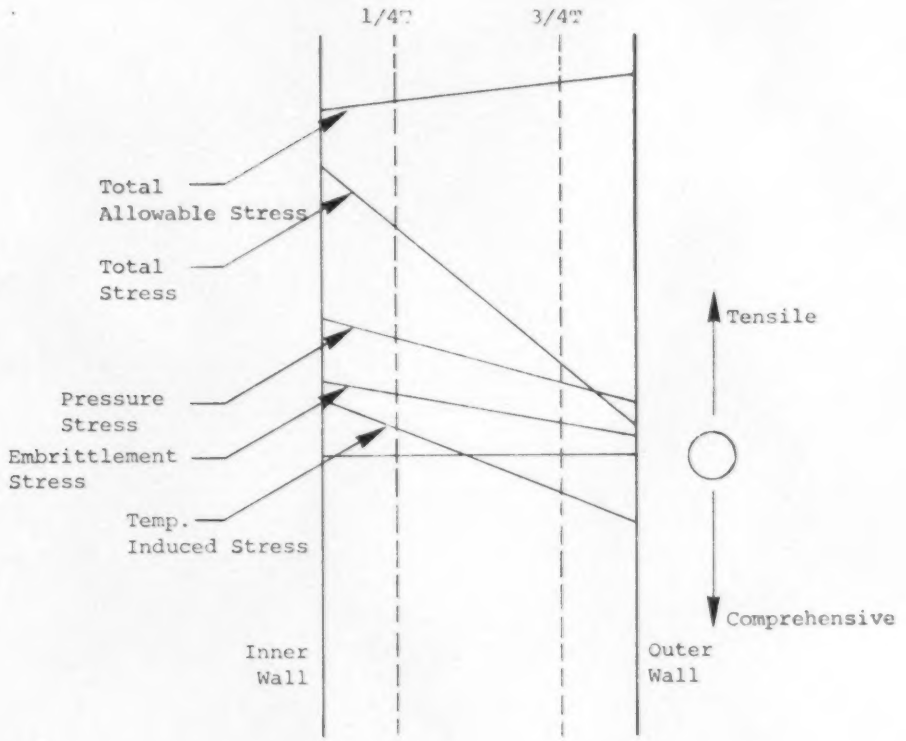


Figure 30-7-3. Cooldown Stress Profile

Stresses Versus Material Properties

The foregoing discussion pointed out that pressure and temperature gradients introduce stresses on the reactor

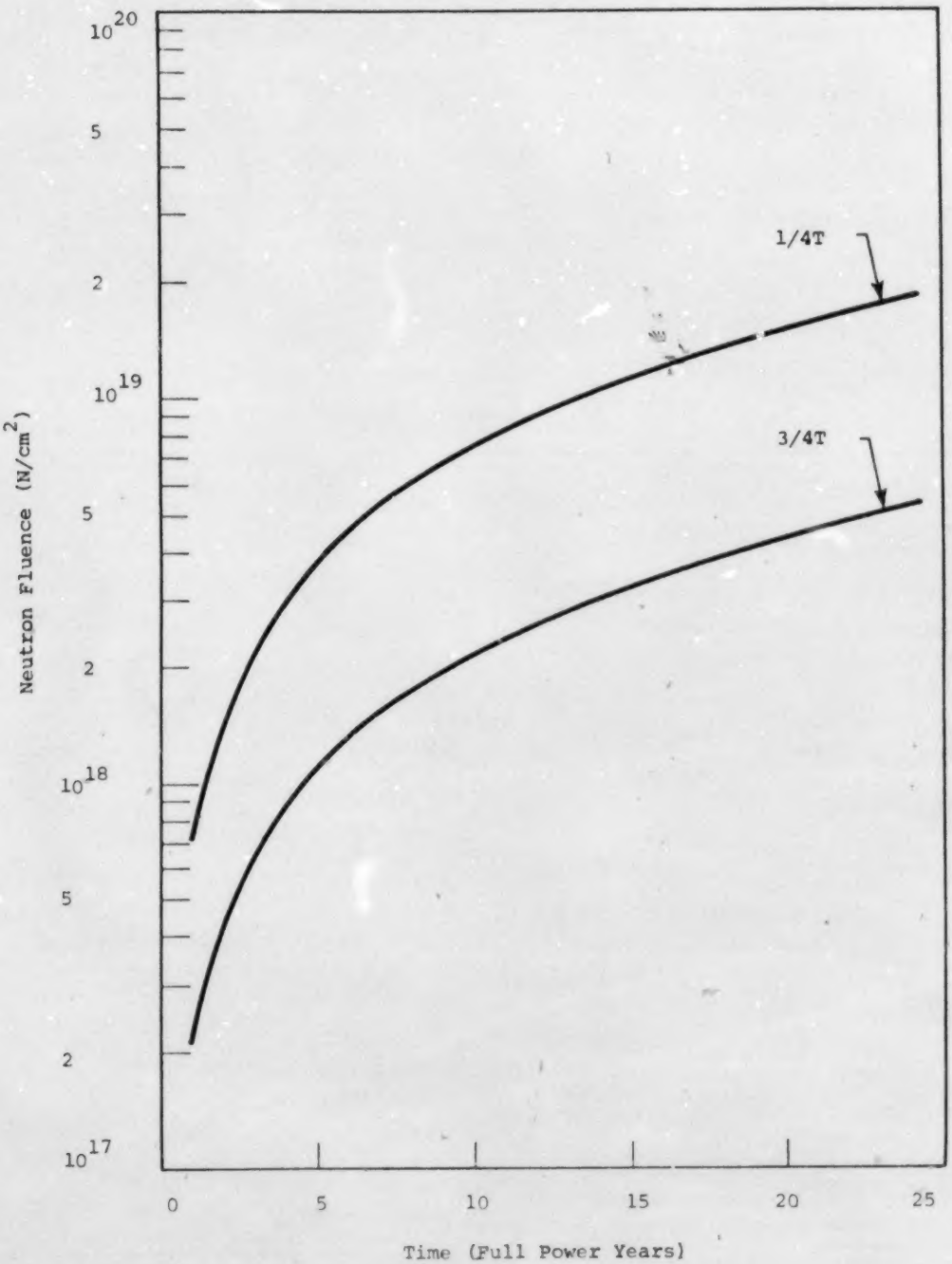


Figure 30-7-4. Typical Vessel Fluence at 1/4T and 3/4T as Function of Full Power Service Years (4 Loop Plant)

vessel. Stresses are categorized as primary, secondary, and local. Primary stress is a stress developed by the imposed loading. Secondary stress is a stress developed by the self constraint of the structure. Local (or peak stress) is the highest stress in the region under consideration. The basic characteristic of a local stress is that it causes no significant distortion but is objectionable because it may be a source of fatigue failure, brittle fracture, or stress corrosion cracking.

Several strength theories of failure have been developed based upon these types of stresses. Their purpose is to predict when failure will occur, assuming that the material behavior in a simple tension test is known and that failure is considered to be either yielding or rupture. Possible modes of metal failure are:

- excessive plastic deformation
- high strain fatigue
- corrosion fatigue
- stress corrosion
- brittle fracture
- excessive elastic deformation.

Of these failure modes, plastic deformation and brittle fracture are of primary concern for reactor vessels.

Elasticity of materials is an important consideration in choosing material to be used in a reactor vessel. Elasticity is that property of a material which measures ability to return to original shape or size after having been stretched, compressed, or deformed. A material which has a large amount of elasticity would be a rubber ball which can be compressed or stretched to certain limits and still return to its original shape. A material such as lead has very little elasticity because it takes very little force to cause a permanent change in its shape.

Elasticity is measured by the ratio of the stress called

into play within the material by the action of deforming forces to the strain or changes in shape or dimensions. The measurement is the modulus of elasticity.

Formula 30-7-1

$$\text{Modulus of Elasticity} = \frac{\text{Stress}}{\text{Strain}} = \frac{\text{Force/Unit Area}}{\text{Elongation/Length}}$$

Stress is the force applied to an object per unit area. For example, a 1000 lb. force of tension on a steel bar 2 inches on a side would impose a stress of 1000 lbs/4 in^2 or 250 lbs/in^2.

Strain is a measure of the deformation of a body under stress. In the previous example, a 1000 lbs. force resulted in a 250 $lb./in^2$ stress. If this same steel bar were 10 inches long before the stress were applied and 10.15 inches long during the time in which the stress were applied, it would have experienced a strain of 0.15 in/10 or 0.015 in/in .

These two properties, stress and strain are used to construct a diagram by which the properties of a material may be examined. This diagram is called a stress-strain diagram. The stress-strain diagram is used to determine different elastic points possessed by a material under a stress. A typical diagram is shown in Figure 30-7-5.

Referring to the stress-strain diagram, the plot from O to B is a straight line, which indicates that stress is proportional to strain for small strains. This behaves according to Hooke's Law which states that: "The deformation of an elastic body is directly proportional to the magnitude of the applied force, provided the elastic limit is not exceeded." The elastic limit is reached at B after stresses greater than that at point B have been attained. This means that the metal no longer regains its

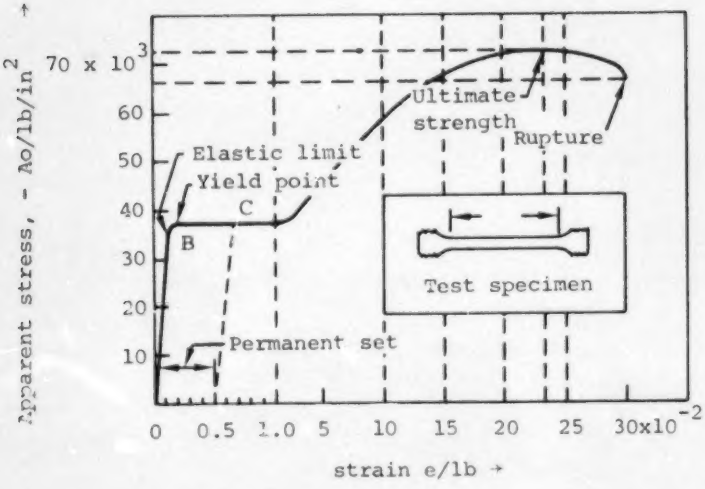


Figure 30-7-5. Typical stress-strain diagram for medium steel in tension. Note the change in scale of strain by a factor of 10 at the broken line.

original length when the distorting forces have been removed. If the tension is removed at C, the strain retraces the broken curve and the bar is left with a permanent set (plastic deformation). As the stress is increased beyond the elastic limit, the yield point is soon reached. At the yield point, a large increase in strain occurs with almost no increase in stress. In fact, when the yield point is reached, the bar appears to flow like an extremely viscous liquid. As the strain is further increased, the stress changes as shown in the figure until rupture occurs. The ultimate tensile strength is the highest point of the curve or the maximum apparent stress that the material can withstand. The ultimate strength is frequently slightly higher than the stress at rupture and may be much above the yield point, as is the case with steel.

It has been found that the yield point, ultimate stress, rupture point etc. are not constant, but rather are a function of the temperature. In fact there is a

point (in temperature) below which almost all materials fail brittlely rather than ductilly as in the foregoing discussion. Brittle failure or fracture is one in which there is a failure of the material without any prior plastic deformation. An example of brittle fracture is glass breaking at room temperature.

Normal steel and other structural type metals have a great deal of plastic deformation before they fail. However, there has been a history of metal containers and structures which have failed as a result of brittle fracture.

In Boston, Mass. in 1919, a 2,300,000 gallon molasses tank fractured on a cold day and twelve men and four horses lost their lives in the disaster. This incident was isolated and it wasn't until World War II when interest in brittle fracture was greatly stimulated when 250 U.S. ships experienced brittle fracture in various degrees of severity. Nineteen of these ships broke in half. In each incident, the temperature was about 40° F.

When tensile specimens are tested at lower and lower temperatures, there exists a temperature at which the yield strength equals the ultimate strength. This is shown as NDT in Figure 30-7-8. At NDT, there is no measurable ductility. The point of Figure 30-7-6 labeled as NDT is called the nil ductility temperature, which is symbolized as T_{NDT}. Metallurgical variables, such as grain size, affect T_{NDT}. Therefore, testing is required to determine the T_{NDT} of a given material. The nil ductility temperature determined by testing is referred to as "reference temperature for nil ductility" and is symbolized by RT_{NDT}.

Brittle fracture is characterized by individual grains of the metal fracturing by "cleavage" of the crystal planes of the metal. Brittle fracture may start at nominal levels of stress provided certain other conditions are satisfied such as the following:

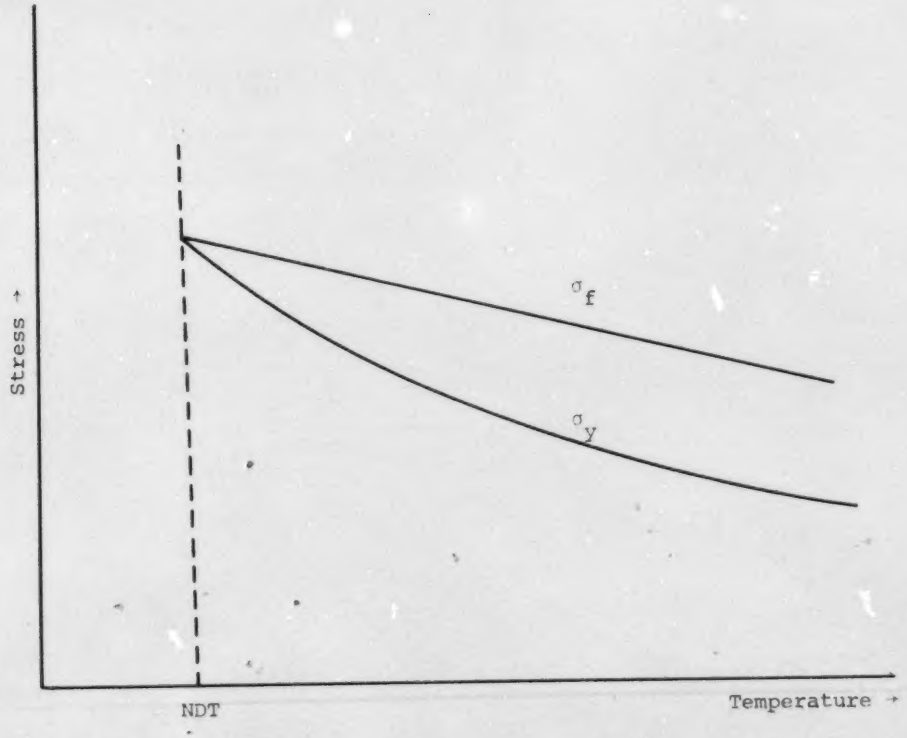


Figure 30-7-6. Stress vs. temperature curve showing plot of failure strength (σ_f) and yield strength (σ_y).

- a flaw such as a crack is present.
- a local stress of sufficient intensity to develop a small deformation at the crack tip is applied.
- a temperature exists which is low enough to promote cleavage fracture of the deformed metal crystals.

Neutron irradiation of the reactor vessel produces damage to the metal's crystalline lattice structure, which increases the yield strength of the material, but also increases the temperature at which brittle failure will occur. Because of this, the RT_{NDT} of the reactor vessel will

increase over core life. An increase of 250°F is not uncommon. Figure 30-7-7 shows this variation and shows that it is also a function of the copper content of the carbon steel reactor vessel.

Since brittle failure must be avoided at all cost, the stresses imposed on the vessel from pressure and temperature gradients must be limited to a safe value. These considerations are used to establish temperatures above which the system may be pressurized and to establish heatup and cooldown curves for the reactor coolant system.

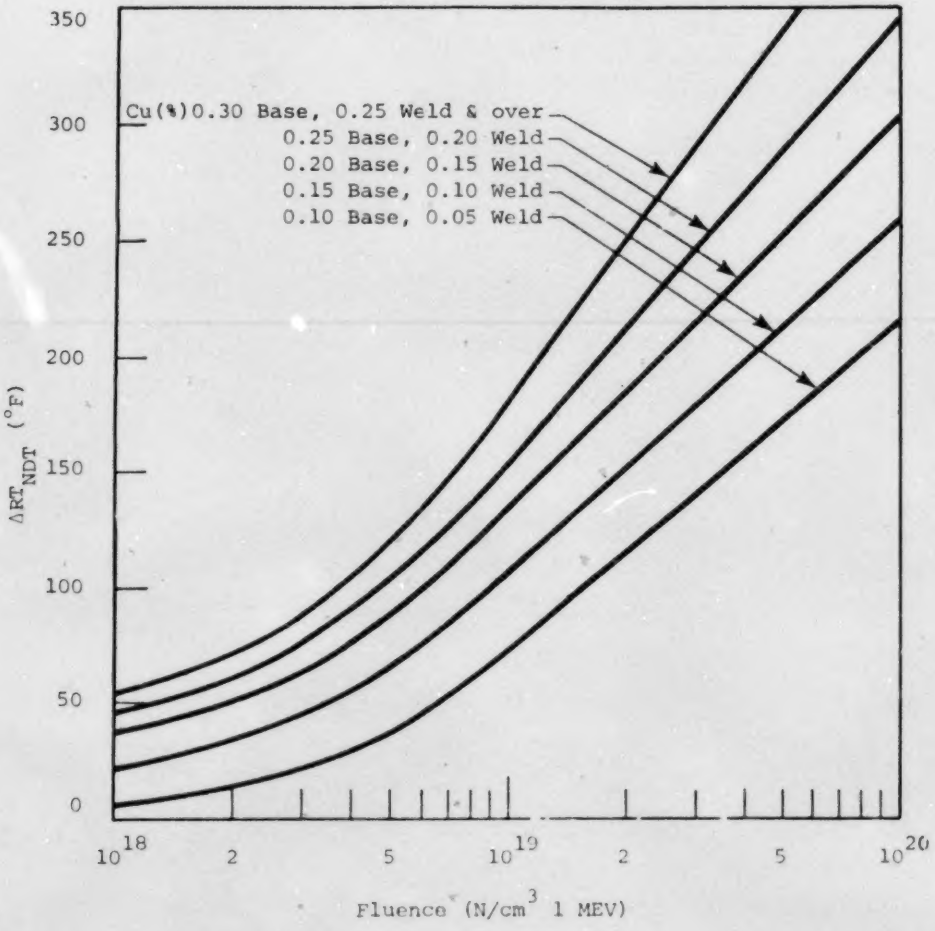


Figure 30-7-7. Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to 550°F Temperature.

Operational Limits

Limits are imposed on reactor operation such that pressure and temperature induced stresses do not cause either plastic deformation or brittle fracture. These limits include allowance for the shift in RT_{NDT} caused by radiation embrittlement and also include a liberal safety margin. These limits are based on the existance of allowable flaws at 1/4T. Thus, if only allowable flaws are present and if temperature and pressure limits are observed, safe operation of the reactor vessel is assured.

Heatup and cooldown limits are applied specifically to avoid brittle fracture. These limits ensure that temperature is raised above RT_{NDT} prior to fully pressurizing the system.

A typical curve which establishes heatup limits for a PWR vessel is illustrated in Figure 30-7-8. Operation must be such that coolant pressure is not increased above the curve. This ensures that temperature is increased well above RT_{NDT} prior to fully pressurizing the system. The criticality limit shown in Figure 30-7-8 takes into account the maximum shift in RT_{NDT} due to radiation plus a liberal safety margin. The reactor core is not permitted to be taken critical until system temperature is above the criticality limit.

A typical cooldown curve for a PWR vessel is illustrated in Figure 30-7-9. As the reactor system is cooled, pressure must be lowered to remain under the curve.

Note, both Figure 30-7-8 and 30-7-9 are valid for 10 effective full power years (EFPY). Nominally a plant is designed for 40 years of service of which about 80% of the time it is "effectively" at 100% power. Hence a nuclear plant would normally operate for about 32 EFPY.

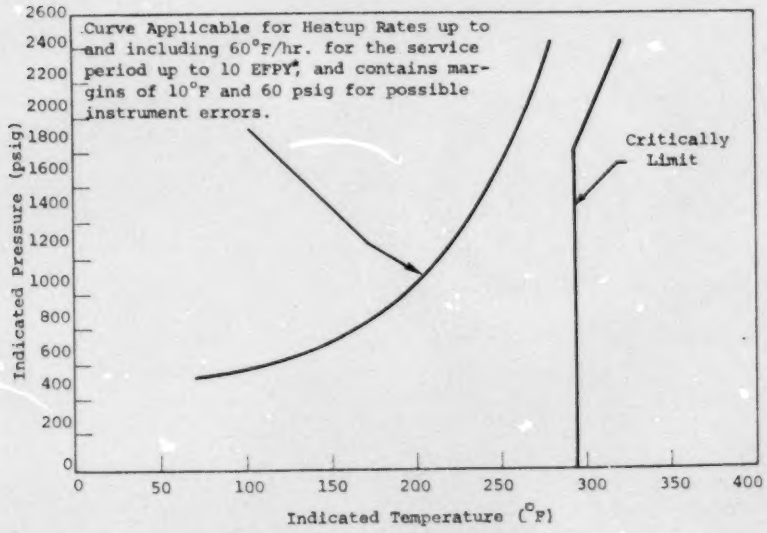


Figure 30-7-8. Reactor Coolant System-Heatup Curve (Typical)

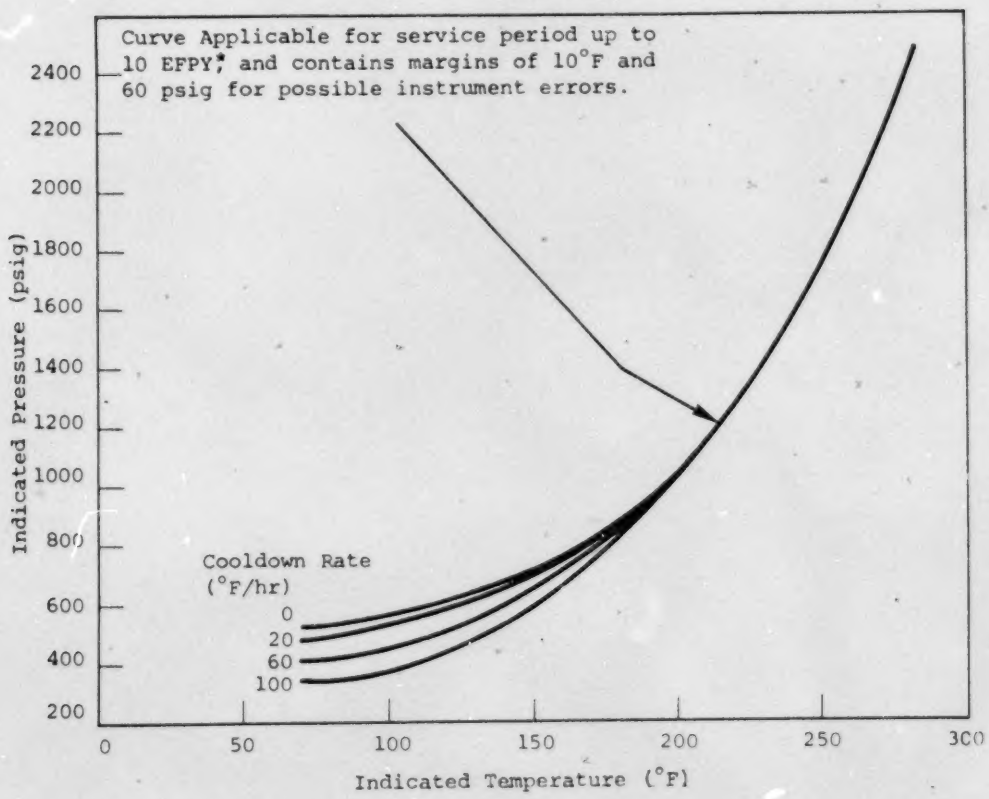


Figure 30-7-9. Typical Reactor Coolant System Cooldown Limitations.
*Note: EFPY refers to effective full power years.

DESIGN AND FABRICATION

Design

Reactor vessels are designed in accordance with provisions for Class I components of Section III of the ASME Boiler and Pressure Vessel Code. Class I components have the most stringent requirements applied. The design is documented in the form of a reactor vessel specification and associated drawings. The adequacy of the design is confirmed by stress analyses which analyze and report ability of the design to withstand static, dynamic, and thermal stresses including after irradiation. The reactor vessel specification and associated design drawings are used as procurement documents in the purchase of reactor vessels. These documents specify:

- materials to be used
- dimensions of overall size, wall thicknesses, and size and form for interfacing features such as nozzles, ledges, closure assembly, mechanism houses, instrumentation connections, lifting lugs, and support mounts.
- cladding material
- closure head assembly features
- special processing requirements such as welding and heat treatment
- destructive testing requirements such as tensile and impact testing.
- nondestructive testing requirements such as radiography and magnetic particle
- acceptance criteria for materials, processing, inspection and testing.

Suppliers of reactor vessels are required to strictly adhere to the requirements of the procurement documents and

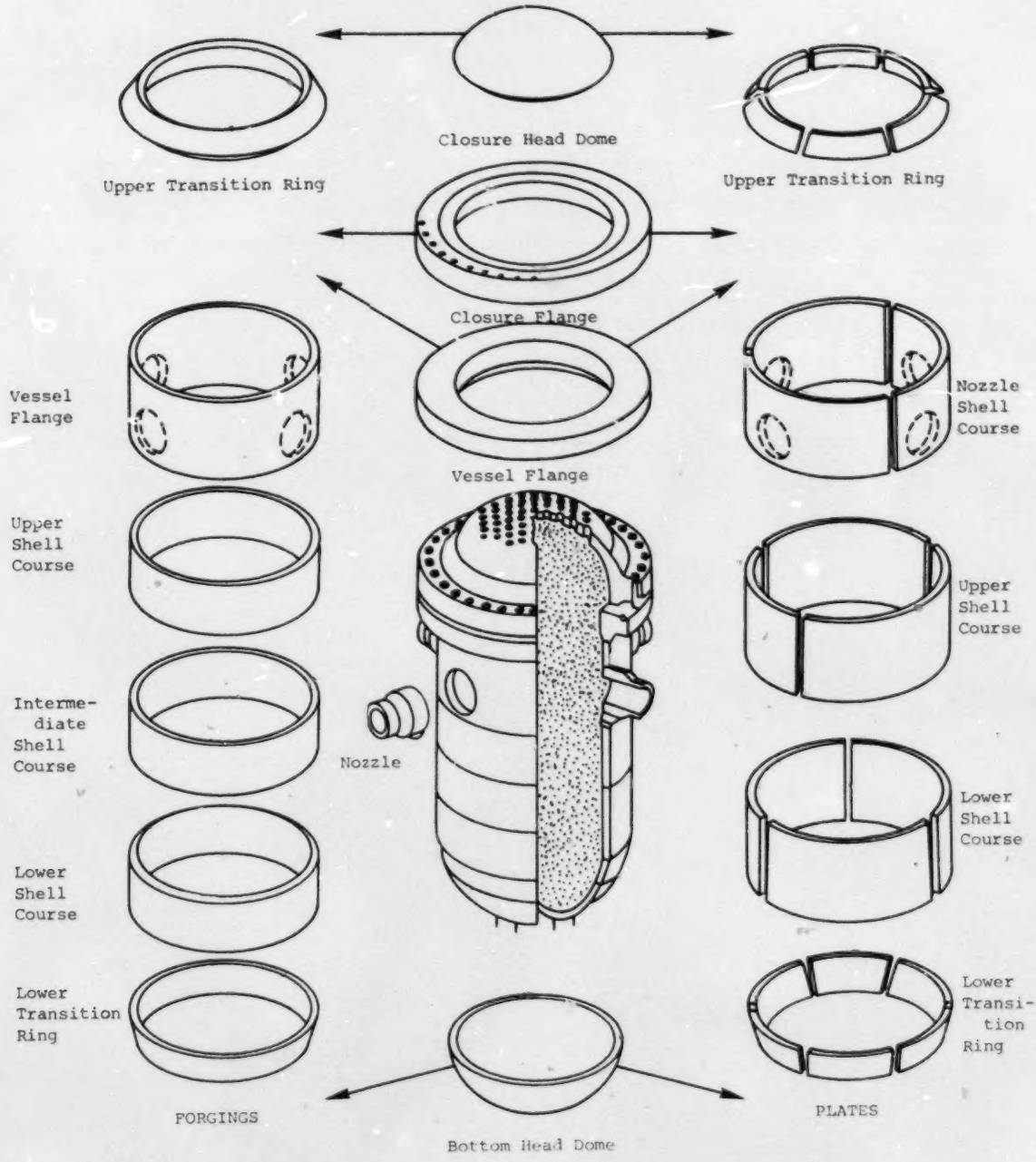


Figure 30-7-10. PWR Reactor Vessel Assembly Showing Forging and Plate-type Construction

to obtain customer approval of deviations therefrom. The supplier is also required to maintain and implement a quality assurance program which ensures compliance with procurement documents. The purchaser is responsible for evaluating the effectiveness of the supplier's quality assurance program and for ultimate acceptance of the reactor vessel.

Basic Fabrication

The reactor vessel is fabricated from a series of cylindrical sections called "shell courses" and from two hemispherically shaped heads. A typical PWR vessel is illustrated in Figure 30-7-10.

Referring to Figure 30-7-10, shell courses are typically rolled plate although forgings may be used. The plate segments are longitudinally welded to form a shell course. The shell courses are circumferentially welded to form the overall structure. The top and bottom heads are hemispherical and fabricated from a dome forging, transition ring, and flange. Head penetrations are provided for control rod drive mechanisms and thermocouple connections. The vessel flange is a forging which is welded to the nozzle shell course as depicted in Figure 30-7-10. The interior wall of the reactor vessel is completely clad with either stainless steel or inconel to resist corrosion by the reactor coolant. Critical machining steps are associated with nozzle and penetration locations, flange bolt holes, keyways for alignment of closure head to vessel assembly, flange mating surfaces, and interior support ledges and keyways.

MATERIALS

Material Specifications

The American Society of Mechanical Engineers (ASME) publishes material specifications as Part II of the ASME Boiler and Pressure Vessel Code. Most of these specifications are identical to material specifications published by the American Society for Testing of Materials (ASTM). The reactor vessel design specification and drawings stipulate for each part the applicable material specification and any modifications, additions, or deletions to the material specification for the specific design. The reactor vessel supplier orders materials from melters and metal finishers to these stipulated material requirements which will permit meeting final dimensions as given by design drawings. Each material specification delineates requirements, as applicable, for:

- melting processes
- heat treating
- chemical analysis
- destructive testing
- nondestructive testing
- marking
- certification

The material supplier normally performs all required inspections and tests on the material and reports the results to the reactor vessel fabricator. However, the fabricator may elect to perform any or all material inspections and tests after receipt of the material. In either case, the quality assurance program must ensure that material meets specified requirements prior to releasing the material for fabrication.

Chemical Analysis

Chemical analysis confirms that specified alloying ingredients are present in the correct percentage. Also, where undesirable constituents, such as copper, have imposed limits, chemical analysis confirms compliance with such limits. Each material specification stipulates the chemical requirements to be met by any heat of material to that specification. In addition, the reactor vessel fabricator must include in the order for materials any additional chemical requirements and limitations imposed by the design documents for the specific reactor vessel. An example of chemical requirements based on material specification SA-533 is given in Table 30-7-1. Material processing by the reactor vessel fabricator does not alter the chemical composition of starting materials.

Table 30-7-1

Example Chemical Requirements

Alloy Material	Composition (percent)
Carbon	0.25 max
Manganese	1.15 → 1.50
Phosphorous	0.035 max.
Sulphur	0.040 max.
Silicon	0.15 → 0.30
Molybdenum	0.45 → 0.60
Nickel	0.40 → 0.70
Copper	0.12 max.

Mechanical Tests

Mechanical tests confirm that the strength and ductility requirements imposed by design are met by the materials used to fabricate the pressure vessel. Mechanical tests are destructive in nature and are therefore also

termed destructive tests.

The mechanical properties of metals are affected by chemical constituents, heat treatment, and metal deforming processes such as rolling, bending, and forging. Thus, mechanical tests are performed by the material supplier after all finishing and heat treatment operations have been completed. However, the reactor vessel fabricator will of necessity subject the material to stress relief heat treatment of welded joints. The fabricator may also bend plate sections to form shell course segments. These processes by the fabricator have potential for altering the mechanical properties of materials. Thus, the fabricator must also perform mechanical testing.

Test specimens are taken from a location of 1/4T where T equals material thickness. Specimens are taken from either production material or from separate test pieces which are from the same heat and which receive the same finishing and heat treatment as the production lot. Because of the destructive nature of the tests, the reactor vessel fabricator processes control material samples in conjunction with vessel fabrication. These controlled samples receive the same bending and heat treatment as their corresponding reactor vessel parts. Upon completion of fabrication, specimens are taken from the control samples and tested to verify that mechanical properties are still within limits.

Mechanical test methods include tensile, drop weight, Charpy V-notch, and bend tests. Except for bend tests, these tests are applied to vessel materials and to weld material qualifications. Descriptions of these tests are as follows:

- tensile test - The tensile test is a static test that uses the well known tensile test machine which pulls the specimen until it fails. The machine

uses pressure detectors and strain gages to measure and plot the elongation of the specimen versus the load applied to it. Since the applied load and the cross section of the specimen are known, the yield and ultimate stresses of the specimen are readily determined. The tensile test machine develops the stress-strain curve illustrated in Figure 30-7-5.

- drop weight test - The drop weight test is a dynamic test which determines T_{NDT}. A test specimen having a preplaced crack in a weld buildup or the underside is placed horizontally in a test fixture. The fixture permits a free-falling weight to strike the test specimen. The specimen is cooled to a preselected temperature and drop weight tested. Crack propagation, if any, is ductile above the nil ductility temperature. The test is performed at consecutively lower temperatures of the test specimen until T_{NDT} is determined by the occurrence of brittle fracture.
- Charpy V-notch - The Charpy V-notch test (C_N) is a dynamic test which uses a free swinging weighted pendulum for striking the test specimen. The test specimen is a rectangular bar with a "V" shaped notch. The specimen is mounted such that its notch is on the opposite side of the impact from the pendulum. The specimen is cooled to a temperature not greater than T_{NDT} + 60°F and impact tested. The machine measures the energy absorbed by the specimen and the broken specimen is examined for lateral expansion. If absorbed energy is at least 50 ft-lb and lateral expansion is at least 35 mils, ductile failure has occurred and RT_{NDT} equals T_{NDT}. If absorbed energy or lateral expansion is less than these limits, additional specimens are tested at

consecutively higher temperatures until both limits are met. In such case, RT_{NDT} equals T_{NDT} plus the increase in temperature from the initial test.

Example

Problem #1: Determine RT_{NDT}

Given:

$T_{NDT} = 15^{\circ}F$ as determined by drop weight testing

C_V at $70^{\circ}F$ yields 62 ft-lb absorbed energy and 38 mils lateral expansion

Solution:

$RT_{NDT} = T_{NDT} = 15^{\circ}F$ because C_V limits are met

Problem #2 Determine RT_{NDT}

Given:

$T_{NDT} = 15^{\circ}F$

C_V at $70^{\circ}F$ yields 20 ft-lb and 10 mils

C_V at $75^{\circ}F$ yields 35 ft-lb and 12 mils

C_V at $80^{\circ}F$ yields 52 ft-lb and 36 mils

Solution:

$RT_{NDT} = T_{NDT} + T_{increase} = 15^{\circ}F + (80^{\circ}F - 70^{\circ}F)$

$= 15^{\circ}F + 10^{\circ}F = 25^{\circ}F$

- bend tests - The bend test is a static test which physically bends a rectangular test specimen around a mandrel of known radius. The bent specimen is visually examined (typically at 5x or 10x magnification) for cracks and voids. This test determines ductility and freedom from flaws and it is used for testing welding procedure qualification test pieces.

Heat Treatment

Heat treatment of materials is performed by the

material manufacturer and the reactor vessel fabricator. Heat treatment by the material manufacturer is that required by the material specification. The heat treatment specified is that which will produce the desired mechanical properties in the material. The material manufacturer must include in the material certification the type of heat treatment used and, if specified by the material specification, the temperature and time at temperature seen by the material. The fabricator performs stress relief heat treatments after welding operations. Time-temperature chart recordings are required for each post weld heat treatment.

Nondestructive Examination of Materials

All pressure retaining material is nondestructively examined for unacceptable flaws. In most cases, volumetric examination is required. Volumetric examination is by either radiography (RT) or ultrasonic (UT) examination. Surface examination, when required, is by either magnetic particle (MT) or liquid penetrant (PT) examination. MT is used for magnetic materials and PT for nonmagnetic materials. Table 30-7-2 lists typical requirements.

Table 30-7-2

Typical Nondestructive Examinations

Material Type	RT	UT	MT or PT	
Plate		X		
Weld Repair of Plate	X		X	
Forgings		X	X	
Coarse grain & complex shape forgings	X		X	
Weld Repair of Forgings	X	(1)	X	
Bolting		(2)	(3)	

Notes: (1) UT of weld repair of certain forgings is permitted in lieu of RT
(2) UT of bolting materials greater than 2 inches is required
(3) MT or PT of bolting materials of 1 to 2 inches is required

Weld Repair of Materials

Unacceptable flaws found by nondestructive examination of materials is repaired by welding except that repair of bolting material is not permitted. Repair procedures must include an examination to verify removal of flaws and to accept the completed weld repair. Verification of flaw removal is by MT or PT of the excavated area. Acceptance of the repair weld is by the method listed in Table 30-7-2. The welding repair procedure and the welder must be qualified. Such qualification is based on the testing of qualification specimens which are welded in accordance with the procedure.

Certification by Material Manufacturer

The material manufacture certifies materials by submittal of a Certified Material Test Report. This certification affirms that the contents of the report are correct and accurate and that all test results and operations are in compliance with the material specification and requirements invoked by the material purchase order. Where the material manufacturer subcontracts for starting material or processing a Certified Material Test Report from each subcontractor and a Certified Material Test Report for only the operations performed by the material manufacturer is permissible. While there is no mandatory format, Certified Material Test Reports identify (1) material specification, grade, class, and condition of material, (2) heat or lot number of materials represented by the certification, (3) results of chemistry analysis for each constituent having a specified range or limit, (4) type and results of each required mechanical test, (5) method of heat treatment and, when specified, heat treatment temperature and time (6) radiographs of weld repairs, and (7) other data requested by the purchase order.

Marking of Material

Material is marked by the material manufacturer with the applicable specification number, grade of material and heat or lot number. Except for small products having less than 1 inch in any direction for the application of marking, each piece of material is marked. For small products, the package or container is marked. Such marking permits direct correlation with the Certified Material Test Report representing the material.

FABRICATION PROCESSING

Purpose

The reactor vessel fabricator prepares processing instructions which identify all necessary manufacturing, inspection, and test steps in the proper sequence for the fabrication of the vessel in accordance with design. Such processing is typically prepared by a product engineering department and approved by a quality engineering group. In addition, an outline of the processing may be required by the contract to be submitted to the Purchaser for approval. The processing has the following purposes:

- identify all necessary steps for fabrication, inspections, and tests.
- identify correct sequences for fabrication, inspection and tests.
- permit application of hold points for customer inspections

Typical Processing

Review of Figure 30-7-10 reveals that many parts comprize the reactor vessel. Starting from plates and forgings, the fabricator sizes, forms, welds, heat treats, and machines to arrive sequentially at finished pieces,

subassemblies, assemblies, and the final component. The overall processing is illustrated in Figure 30-7-11.

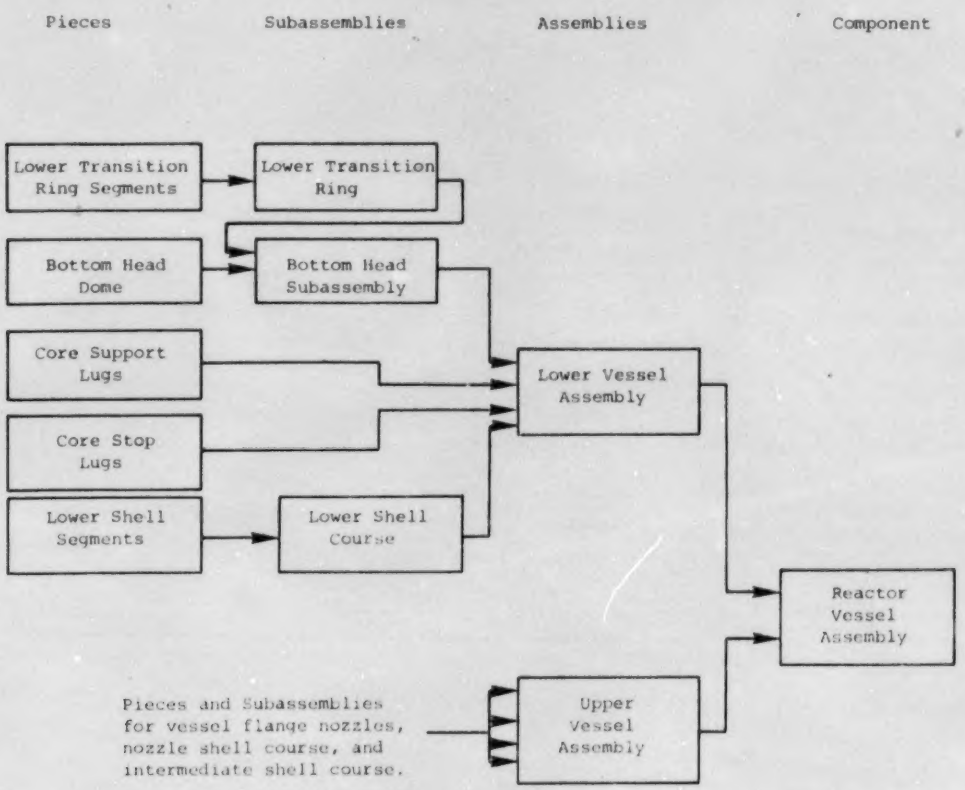


Figure 30-7-11. Major Processing Phases for Vessel Without Closure Head Assembly

Processing instructions are prepared for each block identified in Figure 30-7-11. In this manner, detailed instructions are provided for manufacture, inspection, and test, as required, for each piece, subassembly, assembly, and final component. Such instructions must implement the requirements of the ASME Boiler and Pressure Vessel Code for Class 1 components. Key features of Code requirements are:

- material identity marking is maintained at all times
- preheat is necessary prior to thermal cutting or

welding
- forming processes (e.g. bending plate to form) are qualified by forming of test specimens.
- dimensional and form tolerances of drawings are verified by inspection
- fitting and aligning of parts to be joined by welding is verified by inspection
- welding procedures are qualified
- welders are qualified
- welds are nondestructively examined (mostly by radiography and magnetic particle examination)
- nondestructive examination personnel are qualified and use approved procedures
- weld repairs are made by qualified welding procedures and welders and are examined by qualified persons.
- cleaning steps and inspections for cleanliness are used at key steps such as prior to load in a heat treatment furnace.
- post weld heat treatments are performed by approved procedures and time-temperature recordings are made.
- status of inspections and tests is evident at all times
- the completed component is hydrostatically tested
- the completed component is certified by the fabricator.

Component Hydrostatic Test

The hydrostatic test is the final nondestructive test of the completed reactor vessel. It is a pressurization of the completed reactor vessel at 1.25 times the design pressure. It is performed to confirm the integrity of the reactor vessel and to verify the results of all the other non-destructive tests performed on the vessel. The hydrostatic test is performed at a temperature of 60°F above the measured RT_{NDT} for the reactor vessel.

The fabricator uses blind flanges and plugs to close all openings. Those on high points are fitted with vents to permit venting of all air as the vessel is filled with water. Direct reading pressure gages are installed which have a range of between 1½ to 4 times the test pressure. Pressure gages are calibrated prior to each test or at two week intervals if they are used in a series of tests. Overpressure protection is typically provided by relief valves.

After filling and venting, the water pressure is raised to test pressure by a positive displacement pump. Test pressure is held for a minimum of ten minutes after which visual examination for leaks is performed. During this visual examination, pressure may be lowered to 3/4 of test pressure. Any leak, other than from temporary seals, is cause for rejection.

Component Nameplate and Certification

Reactor vessel fabricators must possess an "N" stamp for Class 1 components. The N stamp is obtained by application to the American Society of Mechanical Engineers (ASME). The ASME reviews and audits the fabricators quality assurance program and issues a certificate of N stamp authorization upon acceptance of the fabricator's quality assurance program. This permits the fabricator to stamp the vessel name plate and to certify the vessel to the Code.

A nameplate is attached to each reactor vessel. This nameplate has sufficient information to permit correlation with the Data Report Form representing the vessel. As a minimum, the nameplate identifies:

- "N" symbol which is N for reactor vessels
- component Class which is Class 1 for reactor vessels
- name of manufacturer

- design pressure and coincident temperature
- manufacturer's N Certificate serial number
- year of manufacture.

Certification of the reactor vessel is made by the fabricator by Data Report Form N-1. This form, which is illustrated in Figure 30-7-12, is supplemented by lists and drawings such that the Data Report provides sufficient information to permit retrieval of all design, fabrication, inspection, and test documents, procedures, and records. The fabricator also attaches Certified Material Test Reports for all pressure boundary materials used in the reactor vessel. These are reports completed by the fabricator for those materials whose properties were affected and determined by the fabrication process, and those submitted by the material manufacturer for materials whose properties were not affected by the fabrication process.

RECEIPT AND INSTALLATION

Owner's Acceptance

The owner of the nuclear power plant ultimately accepts the delivered reactor vessel. Therefore, the owner, i.e., the nuclear utility, establishes a program which develops a basis for acceptance. This program is typically comprized of audits and inspections.

Quality assurance audits by the utility are typically performed of the reactor vessel design agency and the reactor vessel fabricator. In addition, the utility may accompany these contractors on their audits of subcontractors. These audits determine that an adequate quality assurance program is implemented and is effective in assuring that reactor vessel design, fabrication, inspection, and test operations are complying with Code requirements.

FORM N-1 N CERTIFICATE HOLDERS' DATA REPORT FOR NUCLEAR VESSELS*
As required by the Provisions of the ASME Code Rules, Section III, Div. 1

1. Manufactured by _____ (Name and address of N Certificate Holder)

2. Manufactured for _____ (Name and address of Purchaser)

3. Type _____ (Horiz. or Vert.) Kind _____ (Tank, Jacketed, Heat Ex.) Vessel No. (Mfrs. Serial No.) _____ (State & State No.) Nat'l Bd. No. _____ Yr. Built _____

3a. Applicable ASME Code: Section III, Edition _____, Addenda date _____, Case No. _____ Class _____.

Items 4–8 incl. to be completed for single wall vessels, jackets of jacketed vessels, or shells of heat exchangers.

4. Shell: Material _____ (Kind & Spec. No.) T.S. _____ (Min. of range specified) Nominal Thickness ___ in. Corrosion Allowance ___ in. Dia. ___ ft. ___ in. Length ___ ft. ___ in.

5. Seams: Long _____ H.T.[1] _____ R.T. _____ Efficiency _____ %

Girth _____ H.T.[1] _____ R.T. _____ No. of Courses _____

6. Heads (a) Material _____ T.S. _____ (b) Material _____ T.S. _____

Location (Top, bottom, ends)	Thickness	Crown Radius	Knuckle Radius	Elliptical Ratio	Conical Apex Angle	Hemispherical Radius	Flat Diameter	Side to Press. (Convex or Concave)
(a)								
(b)								

If removable, bolts used _____ (Material, Spec. No., T.S., Size, Number) Other fastening _____ (Describe or attach sketch)

7. Jacket Closure _____ (Describe as welded, bolted, etc. If bolted give dimensions, describe or sketch)

Drop Weight _____ Pneumatic
Charpy Impact _____ ft-lb Hydrostatic or } Test
_____ °F. Combination } Pressure _____ psi

8. Design Pressure[2] _____ psi at _____ °F at temp. of _____

Items 9 and 10 to be completed for tube sections.

9. Tube Sheets: Stationary. Material _____ (Kind & Spec. No.) _____ in. Thickness _____ in. Attachment _____ (Welded, Bolted) (Subject to press.)

Floating. Material _____ (Kind & Spec. No.) Dia. _____ in. Thickness _____ in. Attachment _____

10. Tubes: Material _____ (Kind & Spec. No.) O.D. _____ Thickness _____ inches or gage _____ Number _____ Type _____ (Straight or U)

Items 11 to 14 incl. to be completed for inner chambers of jacketed vessels, or channels of heat exchangers.

11. Shell: Material _____ (Kind & Spec. No.) T.S. _____ (Min. of range specified) Nominal Thickness ___ in. Corrosion Allowance ___ in. Dia. ___ ft. ___ in. Length ___ ft. ___ in.

12. Seams: Long _____ (Welded, Dbl., Single) H.T. _____ (Yes or No) R.T. _____ Efficiency _____ %

Girth _____ H.T.[1] _____ R.T. _____ No. of Courses _____

13. Heads: (a) Material _____ T.S. _____ (b) Material _____ T.S. _____ (c) Material _____ T.S. _____

Location	Thickness	Crown Radius	Knuckle Radius	Elliptical Ratio	Conical Apex Angle	Hemispherical Radius	Flat Diameter	Side to Press. (Convex or Concave)
(a) Top, bottom, ends								
(b) Channel								
(c) Floating								

If removable, bolts used (a) _____ (b) _____ (c) _____ Other fastening _____ (Describe or attach sketch)
(Material, Spec. No., T.S., Size, Number)

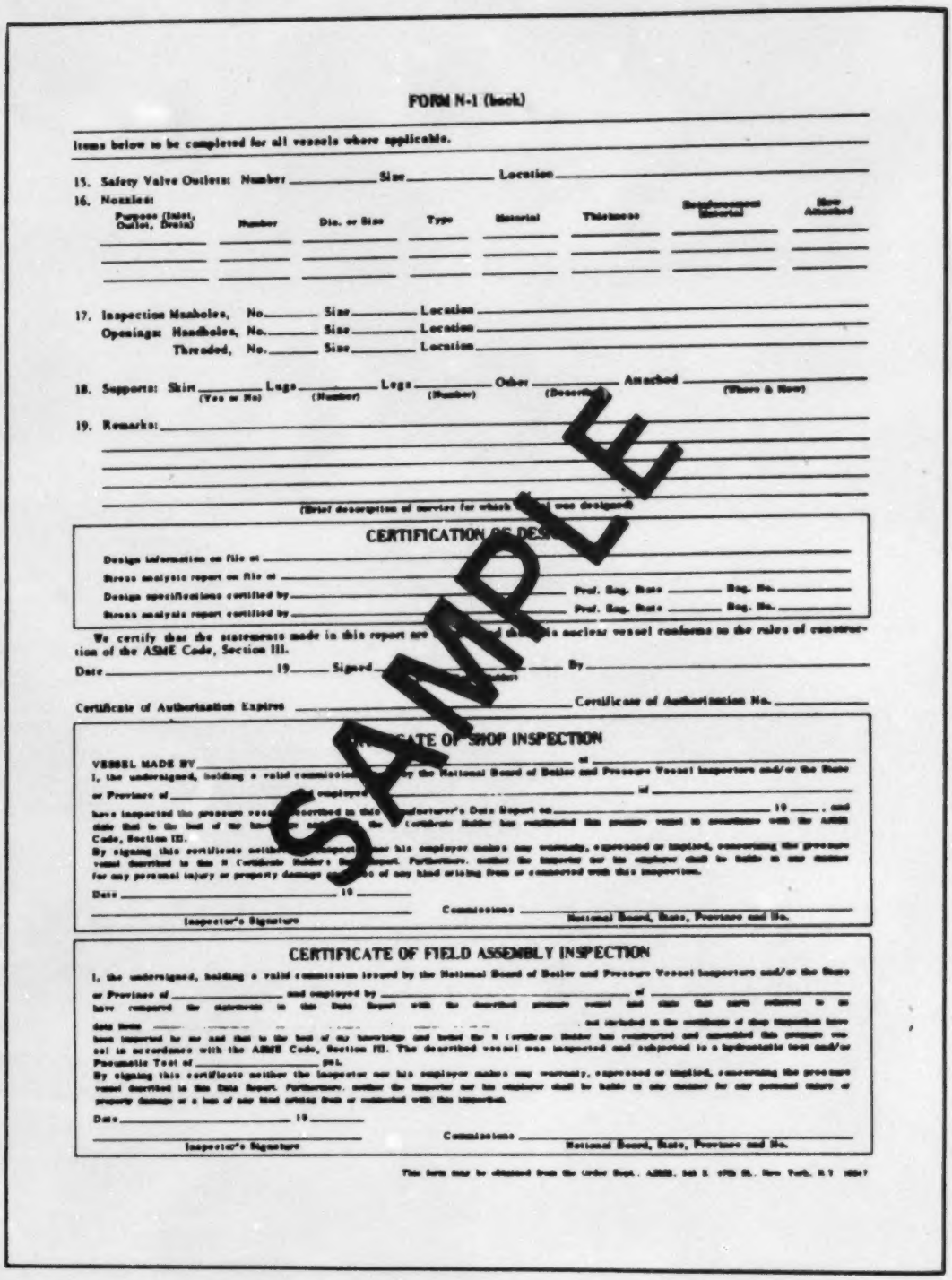
Drop weight _____ Pneumatic
Charpy Impact _____ ft-lb Hydrostatic or } Test
_____ °F. Combination } Pressure _____ psi

14. Design pressure[2] _____ psi at _____ °F at temp. of _____

[1] If Postweld Heat-Treated. [2] List other internal or external pressures with coincident temperature when applicable.
* Supplemental sheets in form of lists, sketches or drawings may be used provided (1) size is 8½" x 11", (2) information on items 1-3 on this Data Report is included on each sheet, and (3) each sheet is numbered and number of sheets is recorded in item 10, "Remarks."

SAMPLE

Figure 30-7-12. Sample Form N-1



FORM N-1 (back)

Items below to be completed for all vessels where applicable.

15. Safety Valve Outlets: Number _____ Size _____ Location _____

16. Nozzles:

Purpose (Inlet, Outlet, Drain)	Number	Dia. or Size	Type	Material	Thickness	Reinforcement Material	Flow Attached

17. Inspection Manholes, No. _____ Size _____ Location _____
Openings: Handholes, No. _____ Size _____ Location _____
Threaded, No. _____ Size _____ Location _____

18. Supports: Skirt _____ (Yes or No) Lugs _____ (Number) Legs _____ (Number) Other _____ (Describe) Attached _____ (Where & How)

19. Remarks: _____

(Brief description of service for which vessel was designed)

CERTIFICATION OF DESIGN

Design information on file at _____

Stress analysis report on file at _____

Design specifications certified by _____ Prof. Eng. State _____ Reg. No. _____

Stress analysis report certified by _____ Prof. Eng. State _____ Reg. No. _____

We certify that the statements made in this report are correct and that this nuclear vessel conforms to the rules of construction of the ASME Code, Section III.

Date _____ 19____ Signed _____ (Certificate Holder) By _____

Certificate of Authorization Expires _____ Certificate of Authorization No. _____

CERTIFICATE OF SHOP INSPECTION

VESSEL MADE BY _____ at _____

I, the undersigned, holding a valid commission issued by the National Board of Boiler and Pressure Vessel Inspectors and/or the State or Province of _____ and employed by _____ of _____ have inspected the pressure vessel described in this Manufacturer's Data Report on _____ 19____, and state that to the best of my knowledge and belief, the Certificate Holder has constructed this pressure vessel in accordance with the ASME Code, Section III.

By signing this certificate neither the Inspector nor his employer makes any warranty, expressed or implied, concerning the pressure vessel described in this Manufacturer's Data Report. Furthermore, neither the Inspector nor his employer shall be liable in any manner for any personal injury or property damage or a loss of any kind arising from or connected with this inspection.

Date _____ 19____

_____ Commissions _____

Inspector's Signature National Board, State, Province and No.

CERTIFICATE OF FIELD ASSEMBLY INSPECTION

I, the undersigned, holding a valid commission issued by the National Board of Boiler and Pressure Vessel Inspectors and/or the State or Province of _____ and employed by _____ of _____ have compared the statements in this Data Report with the described pressure vessel and state that parts referred to as data items _____ and included in the certificate of shop inspection have been inspected by me and that to the best of my knowledge and belief the N Certificate Holder has constructed and assembled this pressure vessel in accordance with the ASME Code, Section III. The described vessel was inspected and subjected to a hydrostatic test and/or Pneumatic Test of _____ psi.

By signing this certificate neither the Inspector nor his employer makes any warranty, expressed or implied, concerning the pressure vessel described in this Data Report. Furthermore, neither the Inspector nor his employer shall be liable in any manner for any personal injury or property damage or a loss of any kind arising from or connected with this inspection.

Date _____ 19____

_____ Commissions _____

Inspector's Signature National Board, State, Province and No.

This form may be obtained from the Order Dept., ASME, 345 E. 47th St., New York, N.Y. 10017

SAMPLE

Figure 30-7-12 Sample Form N-1 (continued)

Inspections are performed at the sites of the designer and fabricator and at the power plant site upon receipt of the reactor vessel. These inspections are planned to address the significant features which are critical to the quality of the reactor vessel. They include the inspection methods of record review, visual inspection, witnessing of processing and testing, and dimensional inspection. Items subject to inspection include:

- reactor vessel specification and design drawing
- stress analysis reports
- certified material test reports
- qualifications of welding procedures and welders
- control material specimens
- heat treatment recordings
- qualifications of nondestructive examination personnel
- inspection and test records
- control of nonconformances
- final interfacing dimensions
- hydrostatic testing
- Data Report Form N-1

Receipt Inspection

Reactor vessels are shipped by rail (or barge) to the nuclear power plant site. The vessels are attached to skids which are fixed to flat bed rail cars. All openings are sealed with temporary plugs and seals to maintain internal cleanliness. Thus, little is available for inspection upon receipt. This is why utility inspection is performed at source of manufacture.

Receipt inspection of reactor vessels typically consists of visual inspection of externals and review of Data Report forms. The visual inspection verifies name plate data and presence and tightness of seals in the various openings and determines if the vessel or its skid has

shifted in transit. If there is evidence of movement between the vessel and its mountings, the cause is corrected and visual inspection is made for evidence of damage to the vessel.

Upon satisfactory completion of receipt inspection, the inspector tags the vessel as accepted and issues an inspection report. However, if a nonconformance is noted, the inspector tags the vessel as nonconforming and issues a nonconformance report.

Installation

The reactor vessel is one of the first components installed in the containment building. Optical tooling may be used for verifying location and form of mounts and for verifying orientation of the vessel on its mounts. Temporary seals are left in place until such time as piping, tubing, and control drive mechanisms are joined to the vessel.

After the primary coolant system has been installed and prior to fuel load into the vessel, a system hydrostatic test is conducted. This test not only tests the reactor vessel but also all connections to it which were made during installation.

The hydrostatic test is performed in essentially the same manner as that performed by the fabricator. However, the system pressure and temperature design limitations apply. Typical limitations for hydrostatic testing of a PWR are given by the pressure-temperature curve in Figure 30-7-13. For the system represented by the figure, coolant temperature must be greater than 135^{o}F prior to pressurizing above 900 psig.

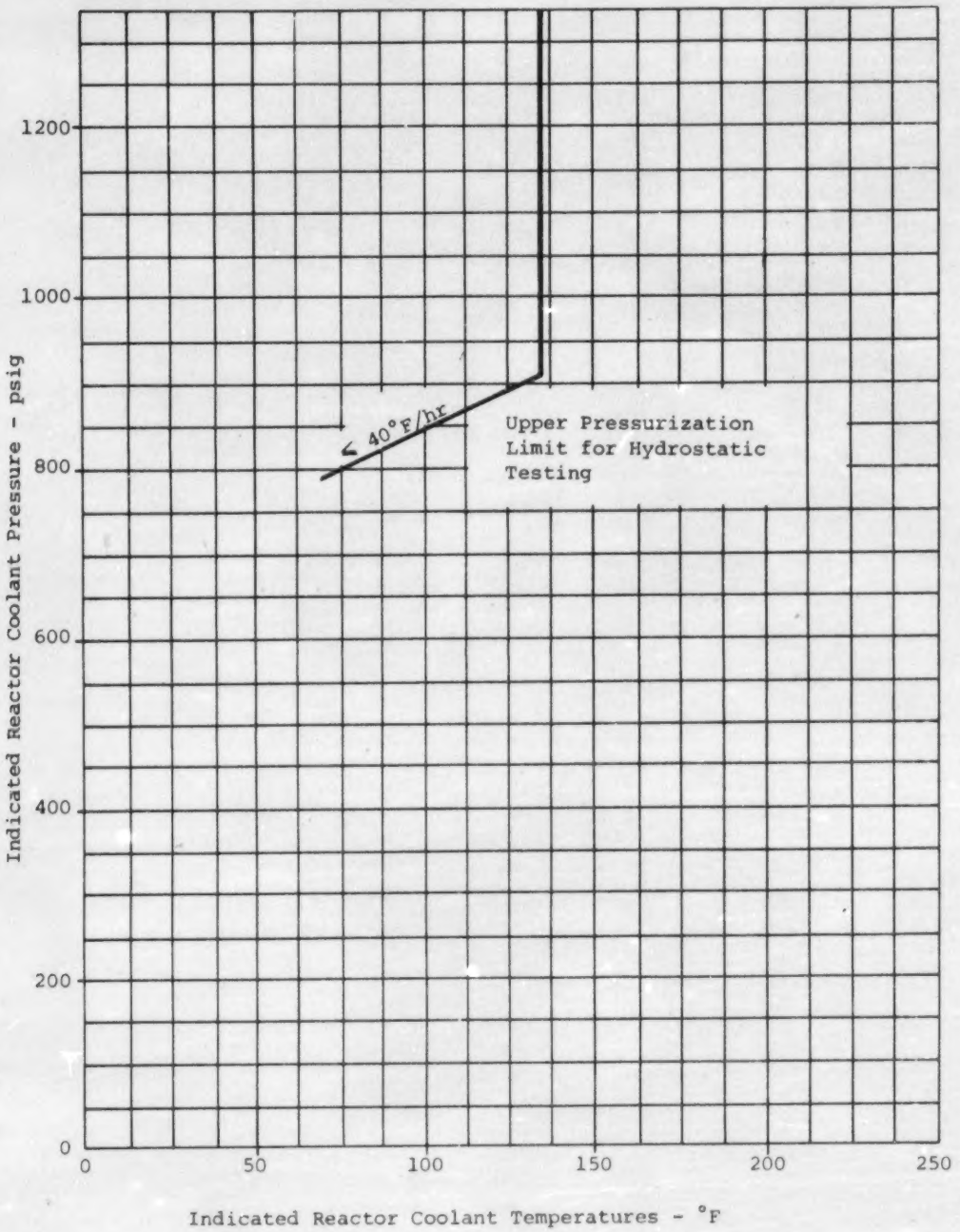


Figure 30-7-13. Hydrostatic Testing Limitations

EXERCISES

The following exercises serve to reinforce the material presented in this module.

A. Perform the following exercises as indicated.

1. Perform one or more of the field trips outlined in the Procedures section of this module.

2. Prepare a report on each of the field trips taken as suggested in the Procedures section of this module.

Answer the following questions as indicated.

3. Explain why the failure or rupture of the reactor vessel could have catastrophic consequences on the reactor core protection and radiation safety of the nuclear power station.

4. Briefly explain how the reactor vessel is constructed. In your description include the following terms:
   a. shell courses
   b. transition rings
   c. head domes
   d. plates
   e. forgings

5. Explain why a low alloy steel consisting of iron alloyed with molybdenum, manganese, and having a low carbon content is used for reactor vessel construction.

6. The various nondestructive testing methods used to inspect the vessel consist of:
   a. Visual Inspection
   b. Magnetic Particle Testing
   c. Dye Penetrant Examination
   d. Radiographic or X-ray Inspection
   e. Ultrasonic Testing
   f. Hydrostatic Testing

   Briefly describe each technique.

B. Answer the following questions as either true or false in the following exercises. An answer key is provided following this set of exercises.

____ 1. Pressure and temperature create, stress on the reactor vessel. So long as the vessel temperature is above its nil-ductility temperature, the vessel will exhibit a normal amount of strain as pressure is applied to it.

____ 2. The steel used for fabrication of the reactor vessel is a low alloy steel consisting of iron alloyed with molybdenum, manganese, and having a low carbon content because this particular alloy is extremely resistant to fast neutron irradiation damage.

____ 3. Brittle fracture can occur when three necessary conditions are satisfied: (1) a flaw such as a crack is present, (2) a stress of sufficient intensity to develop a small deformation at the crack tip is applied, (3) a temperature low enough to promote cleavage is the crystal planes is achieved.

____ 4. Destructive tests which are performed on the reactor vessel materials are: (1) tensile test, (2) Charpy V-notch test, (3) dropweight test, (4) bend test.

____ 5. The tensile test is a static type of test employing a machine which bends a specimen until it cracks.

____ 6. The Charpy V-notch Test is a dynamic test in which a small rectangular beam containing a "V" shaped notch is cooled and struck by a single blow of a pendulum on a point opposite the notch. The amount of energy required to fracture the specimen is determined and this value is plotted as a function of the test temperature.

____ 7. The Dropweight Test is a destructive test using a rectangular beam specimen upon which a pre-placed crack is a brittle weld has been placed. The specimen is then heated to the ΔRT_{NDT} value of $285^{o}F$ (equivalent to 32 effective-full-power years of operation) and supported in a test fixture and struck by a free falling weight to simulate impact loading.

____ 8. The Bend Test is a static test which is primarily used to check welds. These bend tests are performed on weld qualification test pieces to reveal any defects that may be caused by the welding procedure and to provide a measure of joint ductility.

____ 9. RT_{NDT} is the Reference Temperature for Nil Ductility and it is measured by two tests, the Charpy

V-notch Test and the Dropweight Test. The data from these two tests is correlated to obtain the RT_{NDT} value. This temperature is used as a reference temperature for obtaining the heatup and cooldown curves for the reactor pressure vessel.

____ 10. The shift in RT_{NDT} is due to mostly to gamma ray irradiation effects on the pressure vessel steel. It is not unusual for the ΔRT_{NDT} value to be about 250°F for a PWR Vessel.

____ 11. Both heatup and cooldown stresses are comprised of components of pressure stress, thermal stress and embrittlement stress.

____ 12. The 1/4 thickness and 3/4 thickness positions of the reactor vessel represent positions at which the RT_{NDT} value is more conservative. The crystalline lattice structure within the vessel inherently has a somewhat higher RT_{NDT} value than at the surface of the vessel wall. Hence, for a pressure vessel heatup, the 3/4T position is considered most limiting and for cooldowns, the 1/4T position is considered most limiting.

____ 13. The final nondestructive test method is a Hydrostatic Test. This test is a pressurization of the completed reactor vessel at 6.25 times the design pressure. It is performed to confirm the adequacy of the reactor vessel and to verify the results of all the other nondestructive tests performed on the vessel. The hydrostatic test is always run at a temperature above the measured RT_{NDT} value for the reactor pressure vessel.

____ 14. Materials are certified via a Certified Material Test Report which identifies the specification, grade, class, condition, chemistry results, mechanical test results, heat treatment, and NDE of weld repairs.

____ 15. Materials are marked by the material manufacturer to identify specification number, grade, and heat or lot number. The Fabricator maintains this identification throughout manufacture.

____ 16. Weld repair of materials is controlled by NDE verification of removal of flaw, qualified weld repair procedure and welder, and NDE of completed repair by qualified persons.

____ 17. Fabrication processing identifies all necessary steps for fabrication, inspection and test; identifies correct sequence for performing these steps; identifies detailed procedures applicable to these steps; and permits application of Purchaser hold points.

____ 18. The reactor vessel hydrostatic test confirms the integrity of the vessel and verifies the results of all prior NDE.

____ 19. Reactor vessel certification by the fabricator is accomplished via Data Report Form N-1 which identifies pertinent design, material, fabrication, and test information.

____ 20. RT_{NDT} for the reactor vessel is monitored monthly in the plant to ensure that certified material operating procedures may be followed.

The following answer key is for the true/false questions asked in these exercises.

1. T
2. F
3. T
4. T
5. F
6. T
7. F
8. T
9. T
10. F
11. T
12. T
13. F
14. T
15. T
16. T
17. T
18. T
19. T
20. F

MATERIALS

None.

PROCEDURES

The following procedures refer to Exercises #1 and 2 presented in this module.

1. Assign the students/trainees to tour the heating boiler for your facility or one which is readily accessible. Have the students/trainees obtain the inspection, code requirements for that vessel in the heating system.
2. Have the students/trainees visit an electrical power station (either nuclear or fossil) in your area. Have the students/trainees obtain the code requirements for the pressure vessel used at the plant.
3. Have the students/trainees visit an architectural engineering (AE) firm in your locality. Have the students/trainees discuss the role of the AE in pressure vessel installation and inspection. Have the students/trainees examine the ASME Boiler and Pressure Code Book in the AE office, and write a report about the trip.
4. If possible, visit a nuclear plant site where a reactor vessel is being erected. Review the code installation requirements prior to your trip. While at the plant, take notes on the installation procedure and discuss the special precautions required during the installation.

5. Once again, if possible return to the nuclear station when final inspection and acceptance have taken place and review the final acceptance perocedure with QA personnel on site.

DATA TABLE

None.

REFERENCE MATERIALS

ASME, Section III, ASME Boiler and Pressure Vessel Code, American Society of Mechanical Engineers, 1978

GPO, Title 10 of the Code of Federal Regulations Part 50, Appendix G, Fracture Toughness Requirements for Reactor Pressure Vessels, U.S. Government Printing Office, 1978

NBBI, National Board Inspection Code, A Manual for Boiler and Pressure Vessel Inspectors, Pinsky Publishing & Printing Co., Fourteenth Edition, 1977

ASME, Section II, ASME Boiler and Pressure Vessel Code, American Society of Mechanical Engineers, 1978

1. State the primary reason for applying stringent testing and quality control requirements on the design, fabrication, and installation of reactor vessels.

2. The 3/4T location is typically limiting with respect to stresses during heatup because:
   a. Flaws are assumed to be present at 3/4T.
   b. Pressure stresses are tensile and temperature stresses are compressive at 3/4T.
   c. Pressure and temperature stresses are compressive at 3/4T.
   d. Pressure and temperature stresses are tensile at 3/4T.
3. The 1/4T location is typically limiting with respect to stresses during cooldown because:
   a. Flaws are assumed to be present at 1/4T.
   b. Pressure stresses are tensile and temperature stresses are compressive at 1/4T.
   c. Pressure and temperature stresses are compressive at 1/4T.
   d. Pressure and temperature stresses are tensile at 1/4T.
4. The stresses at 1/4T become the limiting factor on operation because:
   a. Flaws are assumed to be present at 1/4T.
   b. Neutron embrittlement is greatest at 1/4T.
   c. Radiation increases tensile strength at 1/4T.
   d. Heatup stresses are greatest at 1/4T.

5. The primary failure modes of concern with reactor vessels are:
   a. Elastic deformation and stress corrosion
   b. Brittle fracture and elastic deformation
   c. Plastic deformation and brittle fracture
   d. Fatigue and stress corrosion.
6. Briefly describe elasticity of metals.

7. Briefly describe the effect on a metal when stress is applied (a) slightly above the yield strength and (b) slightly above the ultimate strength.

8. Briefly describe the effect on the relationship of yield strength and ultimate strength as temperature is reduced.

9. Define nil ductility temperature.

10. Name three conditions which are favorable for brittle fracture to occur.

11. Describe the effect on RT_{NDT} caused by neutron irradiation of the reactor vessel.

12. Describe the basis for establishing heatup and cooldown limits for the operation of a reactor vessel.

13. Name two types of documents used as procurement documents for the purchase of reactor vessels.

14. State the purpose of the following inspections and tests of materials:
    a. chemistry analysis

    b. tensile test

    c. drop weight test

    d. Charpy V-notch test

    e. volumetric nondestructive examination

15. Calculate RT_{NDT} given that T_{NDT} = 20°F; C_V at 75°F failed; D_V at 80°F failed; C_V at 85°F failed; C_V at 90°F passed.

16. Briefly describe the controls applied to weld repair of material flaws.

17. Describe certification of material by a material manufacturer.

18. Describe marking of materials.

19. State four purposes of fabrication processing.

20. Describe typical fabrication controls by indicating the control applicable to each of the following:
    a. material identity

    b. drawing requirements on dimensions and form

    c. welding

    d. nondestructive examination

21. State the purpose of the reactor vessel hydrostatic test and briefly describe conduct of the test.

22. Describe how the fabricator certifies each reactor vessel.

23. Why are owner's inspections mostly performed at source of manufacture of the reactor vessel?